Progress Energy

SEP 1 2 2005

SERIAL: BSEP 05-0119

10 CFR 50.73

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Brunswick Steam Electric Plant, Unit No. 1
Docket No. 50-325/License No. DPR-71
Licensee Event Report 1-2005-005

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc., submits the enclosed Licensee Event Report. This report fulfills the requirement for a written report within sixty (60) days of a reportable occurrence.

Please refer any questions regarding this submittal to Mr. Edward T. O'Neil, Manager – Support Services, at (910) 457-3512.

Sincerely,

B.C. Waldrep

B. C. Waldrep
Plant General Manager
Brunswick Steam Electric Plant

MAT/mat

Enclosure:

Licensee Event Report

IE22

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Brunswick Nuclear Plant
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cc (with enclosure):

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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Brunswick Steam Electric Plant (BSEP), Unit 1	05000325	1 OF 5

4. TITLE
Automatic Reactor Protection System Actuation Due to No Load Disconnect Switch Failure

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
07	13	2005	2005	-- 005 --	00	09	12	2005	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	OTHER Specify in Abstract below or in NRC Form 366A
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Mark A. Turkal, Lead Engineer - Licensing	(910) 457-3066

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	EL	DISC	Delta-Unibus	Y					

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MO	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On July 13, 2005, at 0917 hours, Unit 1 received a Main Turbine trip followed by an automatic Reactor Protection System (RPS) actuation. The cause of the Main Turbine trip was the failure of the B phase of the Main Generator No Load Disconnect Switch (NLDS). Plant systems responded per design. All control rods fully inserted. An expected Reactor Pressure Vessel coolant level shrink resulted in the coolant level decreasing below the Reactor Vessel Water Level - Low Level 1 setpoint, which resulted in appropriate Primary Containment Isolation System (PCIS) isolations. Additionally, coolant level momentarily satisfied the Low Level 2 actuation logic requirements, at which point an additional PCIS isolation occurred and the High Pressure Coolant Injection (HPCI) system initiated but did not inject. Safety/Relief valves B, C, D, and E operated to control pressure. This condition is being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event that resulted in valid actuation of systems listed in 10 CFR 50.73(a)(2)(iv)(B).

The root cause of this event is inadequate design and testing of the NLDS by the vendor; resulting in the NLDS not meeting its nameplate design rating. A temporary modification has been installed on both Unit 1 and Unit 2 which replaced the NLDSs with removable bus bars.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

INTRODUCTION

On July 13, 2005, at 0917 hours, Unit 1 received a Main Turbine [TA] trip followed by a Reactor Protection System (RPS) [JC] actuation. Plant systems responded per design, including the automatic transfer of loads from the Unit Auxiliary Transformer to the Startup Auxiliary Transformer. All control rods fully inserted.

An expected Reactor Pressure Vessel (RPV) coolant level shrink caused the reactor vessel water level to decrease below the Reactor Vessel Water Level - Low Level 1 setpoint, which resulted in a Primary Containment Isolation System (PCIS) [JM] isolation signal to Group 2 (i.e., Drywell Equipment and Floor Drain, Traversing In-Core Probe, Residual Heat Removal (RHR) Discharge to Radwaste, and RHR Process Sample) primary containment isolation valves (PCIVs), Group 6 (i.e., Containment Atmosphere Control/Dilution, Containment Atmosphere Monitoring, and Post Accident Sampling System) PCIVs, and Group 8 (i.e., RHR Shutdown Cooling Suction and RHR Inboard Injection) PCIVs. Additionally, minimum reactor vessel water level momentarily satisfied the Reactor Vessel Water Level - Low Level 2 actuation logic requirements, which caused a Group 3 (i.e., Reactor Water Cleanup System) isolation to occur. The isolation signals closed all of the PCIVs that were open at the time of the actuations.

Safety/Relief valves (SRVs) [RV] B, C, D, and E operated to control pressure.

At 1303 hours, the NRC was notified (i.e., Event Number 41837), in accordance with 10 CFR 50.72(b)(2)(iv)(B), of the Unit 1 RPS actuation and, in accordance with 10 CFR 50.72(b)(3)(iv)(A), of the valid actuation of systems listed in 10 CFR 50.73(a)(2)(iv)(B). This condition is being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event that resulted in valid actuation of systems listed in 10 CFR 50.73(a)(2)(iv)(B).

EVENT DESCRIPTION

Initial Conditions

Unit 1 was in Mode 1, at approximately 100 percent rated thermal power. All required safety-related systems were operable.

Discussion

The direct cause of the Main Turbine trip was the shorting to ground of the B phase of the Unit 1 Main Generator [TB] No Load Disconnect Switch (NLDS) [EL], manufactured by Delta-Unibus Corporation, which electrically connects the generator to the main transformer. This ground resulted in actuation of the generator ground current relay and the generator backup lockout relay, followed by fast closure of the turbine control valves. Turbine Control Valve Fast Closure, Control Oil Pressure - Low (i.e., Function 9 of

EVENT DESCRIPTION (continued)

Technical Specification Table 3.3.1.1-1, "Reactor Protection System Instrumentation") caused the RPS actuation. The generator protection logic and RPS logic functioned as designed. Automatic transfer of loads from the Unit Auxiliary Transformer to the Startup Auxiliary Transformer occurred per design.

As a result of the turbine control valve fast closure and RPS actuation, an expected RPV water level shrink caused reactor water level to momentarily drop to near the Reactor Vessel Water Level - Low Level 2 trip setpoint. All appropriate system isolations and actuations occurred as designed. When RPV water level dropped below the Reactor Vessel Water Level - Low Level 1 setpoint, PCIS Groups 2, 6, and 8 properly actuated. The minimum RPV water level experienced was sufficient to cause a PCIS Group 3 isolation, isolation of the Reactor Building Ventilation [VA] system and actuation of the Standby Gas Treatment [BH] system, tripping of the reactor recirculation pumps [AD], and initiation of the High Pressure Coolant Injection (HPCI) [BJ] system. HPCI did not inject because reactor vessel water level was restored prior to the HPCI injection valve opening. The momentary RPV water level shrink was not sufficient to fulfill the Reactor Core Isolation Cooling (RCIC) [BN] system logic actuation requirements. Subsequent testing confirmed the proper functioning of RCIC instrumentation. RPV water level was maintained using reactor feed pumps [SJ] and the Control Rod Drive (CRD) [AA] system.

The pressure transient caused by the turbine control valve fast closure also caused SRVs B, C, D, and E to open for one to two seconds. Although the peak reactor pressure of 1108 psig was not high enough to reach any SRV setpoint, fast closure of the turbine control valves causes a pressure wave which is expected to be higher in the steam lines than the RPV. This is a known phenomenon which has been experienced at Brunswick as well as other Boiling Water Reactors.

Both plant and Operator response to this event were as expected.

EVENT CAUSE

The root cause of this event is inadequate design and testing of the NLDS by the vendor; resulting in the NLDS not meeting its nameplate design rating. This resulted in the B phase of the NLDS failing when operated at extended power uprate operating conditions.

The NLDS is manufactured by Delta-Unibus Corporation. Each switch assembly consists of a stack of six bus bars, each containing nine contact fingers on each end. The three phase switches are gang operated with a manual crank. The switch rotates on a center point and mates to three corresponding stationary contact plates at each end. The switch was rated, by the vendor, at 25KV with a 25,000 amp forced cooled and 12,500 amp self-cooled continuous current rating. Subsequent to the Unit 1 NLDS failure, the vendor stated that the switch was tested to the 12,500 amp self-cooled continuous current rating. The 25,000 amp rating was based on calculation, assuming a bus duct cooling total air handling output of 15,300 cfm; however, no documentation of this calculation could be produced by the vendor or found on site. The

EVENT CAUSE (continued)

Unit 1 NLDS was installed in April 1995, and subsequently inspected each refueling outage with no signs of significant degradation due to overheating.

In a report dated August 27, 2001, the same vendor evaluated the iso-phase system [EL] and confirmed the acceptability of operating under the planned extended power uprate conditions.

Prior to extended power uprate, the NLDS operated at approximately 21,000 amps. Bus duct cooling was provided by two fans: one fan operation yielded 17,500 cfm flow and two fan operation yielded 23,000 cfm flow. The first phase of extended power uprate increased the load on the switch to approximately 22,860 amps and required the operation of two bus duct cooling fans. The final phase of extended power uprate increased the load on the NLDS to approximately 24,300 amps and also increased the capability of the bus duct cooling system to 34,000 cfm with two fans operating. Subsequent to the Unit 1 NLDS failure, Unit 2 bus duct cooling air flow measurements were performed and determined to be sufficient to support vendor design assumptions for the NLDS. The Unit 1 iso-phase bus and bus duct cooling are sufficiently similar that the Unit 2 measurements are also applicable to Unit 1.

Based on the fact that the NLDS switch was being operated at less than the continuous rating of 25,000 amps and the provided forced air bus duct cooling exceeded the vendor design assumptions, it is concluded that the NLDS did not meet its nameplate rating.

CORRECTIVE ACTIONS

A temporary modification has been installed on both Unit 1 and Unit 2 which replaced the NLDSs with removable bus bars. The ability to remove the bars within one hour, consistent with Brunswick commitments with respect to General Design Criterion 17, "Electric power systems," was confirmed.

Permanent corrective actions, which maintain the ability to implement the delayed source of off-site power (i.e., Unit Auxiliary Transformer Backfeed) within one hour, are currently being investigated and are planned to be completed during the B116R1 refueling outage (i.e., the March 2006, refueling outage for Unit 1) and during the B218R1 refueling outage (i.e., the March 2007, refueling outage for Unit 2).

SAFETY ASSESSMENT

The safety significance of this condition is considered minimal.

The failure of the NLDS caused an automatic RPS actuation; however, during this event all systems functioned as expected, resulting in low safety significance. The NLDS is not a required active component to implement the delayed source of off-site power (i.e., Unit Auxiliary Transformer Backfeed), but it must be opened to facilitate the lineup. In this event, the failure caused the switch to stick closed and mechanical

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(1-2001)

LICENSEE EVENT REPORT (LER)

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant (BSEP), Unit 1	05000325	2005	-- 005 --	00	5 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

SAFETY ASSESSMENT (continued)

assistance was required to open the switch. However, this activity was completed in a relatively short time (i.e., approximately 1 to 1.5 hours). Therefore, if the backfeed lineup had been required, the necessary NLDS lineup could have been achieved in relatively short order.

PREVIOUS SIMILAR EVENTS

A review of events which have occurred within the past three years has not identified any previous similar occurrences related to inadequate vendor design.

COMMITMENTS

No regulatory commitments are contained in this report. Those actions discussed in this submittal will be implemented in accordance with corrective action program requirements.